ENCLOSURE 1

REVISED SAFETY EVALUATION OF FRAMATOME COGEMA FUELS TOPICAL REPORT BAW-10186P "EXTENDED BURNUP EVALUATION"

1 INTRODUCTION

In a letter dated November 24, 1992, from J. H. Taylor, Babcock & Wilcox Nuclear Technologies (B&WNT), to the U.S. Nuclear Regulatory Commission (NRC), B&WNT submitted a Topical Report BAW-10186P, "Extended Burnup Evaluation," for NRC review. By letter dated July 19, 1995, B&WNT requested that the review be extended to include a change in the fuel rod power history uncertainty used in TACO3 licensing analyses. Since that time B&WNT has become Framatome Cogema Fuels (FCF).

BAW-10186P describes an improved extended burnup methodology that FCF intends to apply for fuel reload applications. The purpose of this improved methodology is to extend the analysis to a slightly higher burnup range than the previously approved range for different fuel designs. Additional material including responses to the NRC's requests for additional information was submitted by letters dated August 22, and December 6, 1995, June 26, 1996, and January 23, 1997.

The staff reviewed the topical report and the related documents, and approved BAW-10186P in a letter, including a safety evaluation (SE), from D. B. Matthews (USNRC) to J. H. Taylor (FCF) dated April 29, 1997. The NRC staff was supported in this review by its consultant, Pacific Northwest National Laboratory (PNNL). Our consultant's technical evaluation report (TER), which was attached, provided technical findings relative to the review. Subsequently, FCF published an approved version of the report BAW-10186P-A on June 12, 1997.

During the implementation of BAW-10186P-A, FCF raised a question about the limitations on the predicted cladding corrosion levels. FCF, NRC staff, and its contractor reviewer at PNNL held several telephone conferences to reach agreement on the interpretation of the limitations. FCF submitted two letters dated August 29 and October 28, 1997 from J. H. Taylor to USNRC to clarify the corrosion issue. The staff determined that the SE should be revised and reissued to avoid confusion in the future. Thus this revised SE will supersede the SE dated April 29,1997.

2 EVALUATION

The staff reviewed the enclosed TER, and concluded that the TER provides an adequate technical basis to approve BAW-10186P. The staff agrees with PNNL's conclusion that the improved methodology described in BAW-10186P is acceptable for fuel reload licensing applications. Based on our review, the staff adopts the findings in the attached TER. In addition the staff provides an assessment of corrosion limit in the following.

2.1 Oxidation and Crud Buildup (TER Section 3.0(E))

In a letter dated October 28,1997 from J. H. Taylor to USNRC, FCF stated that the predicted oxide/corrosion layers are limited to 100 microns for normal operation and anticipated operational occurrences (AOOs). This limit of oxide and corrosion depth is intended to address the concern of potential ductility reduction and other adverse effects on the cladding integrity for high burnup operations. This limit of 100 microns has been widely used in the industry for fuel rod designs. Thus, the staff considers that the limit of 100 microns for oxide/corrosion including the crud buildup is acceptable.

FCF further proposed a lead test assembly (LTA) program to continue collecting corrosion data during high burnup operations. The LTA program allows a total of eight fuel assemblies in each fuel cycle from different sub-batches to operate even though the predicted corrosion is greater than 100 microns. These assemblies will be designated as lead corrosion assemblies. Typically these assemblies will be placed in non-limiting core positions but with relative high powers to be able to simulate typical operation conditions. Corrosion measurements will be performed after these assemblies are discharged from the core. In any fuel core the total number of LTAs (lead corrosion assemblies plus other LTAs) will not exceed twelve. The staff reviewed the LTA program and determined that this LTA program satisfies the intent of the LTA programs as described in the Standard Review Plan (SRP) 4.2. Therefore, the staff approves the FCF's LTA program.

FCF will use the COROS02 corrosion model for best estimate calculations of corrosion. Best estimate models are used throughout the industry. While the staff recognizes that the corrosion data base has large uncertainty, and different measurement techniques can produce very different results, the staff considers that the use of a best estimate calculation for corrosion analysis is not unreasonable. FCF will continue assessing the corrosion model conservatism to ensure that the best estimate model is consistent and unbiased through high burnups. The NRC consultant PNNL has reviewed the COROS02 corrosion model and found it acceptable as described in the attached TER. Thus, the staff approves the use of a best estimate calculation in the corrosion model.

3 CONCLUSIONS

The staff has reviewed the FCF's extended burnup methodology described in BAW-10186P, and finds that the improved methodology is adequate and thus acceptable for fuel reload licensing applications subject to the following conditions to which FCF has agreed (References 6 and 7).

1) This methodology is acceptable for Mark-B fuel design up to 62 GWd/MTU rod average burnup.

2) This methodology is acceptable for Mark-BW fuel design up to 60 GWd/MTU rod average burnup.

3) This approval does not cover extended burnup operation of Mark-C fuel design.

4) The maximum predicted oxide thickness will be 100 microns.

5) Up to eight fuel assemblies from different sub-batches in each fuel cycle may have fuel rods with predicted oxide layers greater than 100 microns and will be designated as lead corrosion assemblies.

6) The total number of lead test assemblies (lead corrosion assemblies and other LTAs) in any fuel cycle will not exceed twelve.

In addition, as was stated in the TER, the NEMO code calculational uncertainty for use in the TACO3 fuel performance code for licensing analyses is acceptable.

4 REFERENCES

1. "Extended Burnup Evaluation," BAW-10186P, Babcock and Wilcox Fuel Company, Lynchburg, Virginia, transmitted by letter, J. H. Taylor (BWFC) to U.S. NRC Document Control Desk, dated November 24, 1992.

2. Letter, J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC) dated July 19, 1995.

3. Letter, J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC), JHT/95-88, dated August 22, 1995.

4. Letter, J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC), JHT/95-119, dated December 6, 1995.

5. Letter, J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC), JHT/96-042, dated June 26, 1996.

6. Letter, J. H. Taylor (Framatome Cogema Fuels) to NRC Document Control Desk, " Application of BAW-10186P," JHT/97-7, dated January 23, 1997.

7. Letter, J. H. Taylor (Framatome Cogema Fuels) to NRC Document Control Desk, JHT/97-39, dated October 28, 1997.

8. Babcock and Wilcox Fuel Company, March 1993, NEMO-Nodal Expansion Method Optimized, BAW-10180P-A, Rev 1, Babcock and Wilcox Fuel Company, Lynchburg, Virginia.

TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION REPORT OF THE TOPICAL REPORT BAW-10186P (EXTENDED BURNUP EVALUATION REPORT)

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ABBREVIATIONS LIST

AOO	-	Anticipated Operational Occurrence
ASME	-	American Society of Mechanical Engineers
DNB	-	Departure from Nucleate Boiling
DNBR	-	Departure from Nucleate Boiling Ratio
ECCS	-	Emergency Core Cooling System
EOL	-	End of Life
FCF	-	Framatome Cogema Fuels
FGR	-	Fission Gas Release
GDC	-	General Design Criterion
LOCA	-	Loss of Coolant Accident
NRC	-	U.S. Nuclear Regulatory Commission
PCI	-	Pellet Cladding Interaction
PCT	-	Peak Cladding Temperature
PIE	-	Post Irradiation Examination
PNNL	-	Pacific Northwest National Laboratory
RIA	-	Reactivity Initiated Accident
SAFDL	-	Specified Acceptable Fuel Design Limit
SRP	-	Standard Review Plan
TER	-	Technical Evaluation Report

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1.0 INTRODUCTION

Framatome Cogema Fuels (FCF) has submitted to the U.S. Nuclear Regulatory Commission (NRC) a topical report, entitled "Extended Burnup Evaluation," BAW-10186P (Reference 1), for review and approval. This report requests a extension in fuel rod average burnups for their Mark-B (15X15) and Mark-C (17X17) fuel designs for Framatome type reactors, and Mark-BW15 (15X15) and Mark-BW17 (17X17) for Westinghouse type reactors. An additional request was made to extend the scope of this review (Reference 2) to include a change in the fuel rod power history uncertainty used in TACO3 licensing analyses. The original power uncertainty used for TACO3 were based on the calculational uncertainties associated with the FLAME3 neutronics code used at the time the TACO3 code was developed. Since that time FCF has developed the NEMO code (Reference 3) for neutronics and rod power calculations and the calculational uncertainties of the code are lower than for the previous FLAME3 code. This request is evaluated at the beginning of Section 3.0 of this report. This Technical Evaluation Report (TER) will only address the burnup extension of 62 GWd/MTU for Mark B and 60 GWd/MTU for Mark BW designs and the proposed change in the power uncertainties used in TACO3 for licensing analyses (Reference 2). The previously approved burnup extensions have limited the Mark-B fuel designs to a proprietary batch average burnup defined in Reference 5 and the Mark-BW fuel designs up to a lead rod-average burnup level of 60 GWd/MTU (Reference 6). The Mark-C design is not covered in this review because FCF has only a limited amount of performance data for this design and does not currently have an operating reactor utilizing this design.

It should be explained that Framatome Cogema Fuels was previously named the B&W Fuel Company (BWFC) a part of B&W Nuclear Technologies and prior to BWFC was named Babcock & Wilcox (B&W). Some of the references in this TER refer to these different company names depending on the date the reference was generated.

Pacific Northwest National Laboratory (PNNL) has acted as a consultant to the NRC in this review. As a result of the NRC staffs and their PNNL consultants review of the topical report, a list of questions were sent by the NRC to FCF requesting clarification of specific design criteria and licensing analyses (Reference 7). FCF partially responded to those questions in Reference 8 and provided the remaining responses in Reference 9. Following a February 26, 1996 telecon with NRC and PNNL, FCF agreed to supply additional information (Reference 10) to support their request for a burnup extension.

This review was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 11). The objectives of this fuel system safety review, as described in Section 4.2 of the SRP, are to provide assurance that 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), 2) fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) coolability is always maintained. A "not damaged" fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analysis. Objective 1, above, is consistent with General Design Criterion (GDC) 10 (10 CFR 50, Appendix A) (Reference 12), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 (Reference 13) for postulated accidents. "Coolable geometry" means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for the LOCA are given in 10 CFR 50, Section 50.46 (Reference 14).

In order to assure that the above stated objectives are met and follow the format of Section 4.2 of the SRP, this review covers the following three major categories: 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs; 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents; and 3) Fuel Coolability, which are applied to postulated accidents. Specific fuel damage or failure mechanisms are identified under each of these categories in Section 4.2 of the SRP. This TER discusses under each fuel damage or failure mechanism listed in the SRP the FCF design limits, analysis methods and data used to demonstrate that the SAFDLs are met up to the rod-average burnup levels of 62 GWd/MTU for Mark B and 60 GWd/MTU for Mark BW designs.

The purpose of design criteria or limits are to provide limiting values that prevent fuel damage or failure and fuel coolability/control rod insertability for postulated accidents with respect to each mechanism. Reviewed in this TER is whether FCF fuel designs have adequate data to demonstrate that their fuel designs can operate satisfactorily up to rod-average burnup levels of 62 GWd/MTU for Mark B and 60 GWd/MTU for Mark BW designs as defined by the SAFDLs for normal operation, AOOs and postulated accidents.

The Mark B and Mark BW fuel designs are briefly discussed in the following section (Section 2.0). The fuel damage and failure mechanisms are addressed in Sections 3.0 and 4.0, respectively, while fuel coolability is addressed in Section 5.0.

2.0 FUEL SYSTEM DESIGN

The Mark-B design is a 15X15 assembly with Zircaloy spacer grids. The fuel assembly consists of 208 fuel rods, 16 control rod guide tubes, 1 instrumentation tube assembly, 7 segmented spacer sleeves, 8 spacer grids, and bottom and top nozzles. The guide tubes, spacer grids and end nozzles form the structure of the assembly where the fuel rods and tubes are arranged in a 15X15 array. The center position in the assembly is reserved for instrumentation. The structural materials consist of Zircaloy-4 and Inconel except for the axial power shaping rod cladding which consists of stainless steel.

The Mark-BW designs are 15X15 and 17X17 assemblies with Zircaloy grids for Westinghouse type reactor reloads. The 15X15 assembly consists of 204 fuel rods, 20 control rod guide tubes, 1 instrumentation tube assembly, 8 spacer grids, and top and bottom nozzles with the holddown spring being a leaf configuration. The 17X17 assembly consists of 264 fuel rods, 24 control rod guide tubes, 1 instrumented tube assembly, 8 spacer grids, and top and bottom nozzles with the holddown spring being a leaf configuration.

3.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation including AOOs. The evaluation portion of each damage mechanism evaluates the analysis methods, analyses and data used by FCF to demonstrate that their design criteria are not exceeded during normal operation including AOOs for their fuel designs up to rod-average burnup limits of 62 GWd/MTU for Mark B and 60 GWd/MTU for Mark BW designs.

A request was made by FCF to extend the scope of this review (Reference 2) to include a change in the fuel rod power history uncertainty used in TACO3 licensing analyses. The TACO3 code is used by FCF in many of the analysis methods discussed below to verify that the design criteria in this section and Sections 4.0 and 5.0 are met. The original power uncertainty used for TACO3 licensing applications were based on the calculational uncertainties associated with the FLAME3 neutronics code used at the time the TACO3 code was developed. Since that time FCF has developed the NEMO code for neutronics and rod power calculations and the calculational uncertainties of the code are lower than for the previous FLAME3 code. The NEMO neutronics code and calculational uncertainties have been approved by the NRC (Reference 3) for analysis of fuel powers and neutronics. Therefore, the use of the NEMO calculational uncertainty for use in TACO3 licensing applications is considered to be acceptable.

(A) STRESS

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria for cladding stress should ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The FCF design criteria for fuel rod cladding and assembly stresses are based on guidelines established in Section III of the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (Reference 15). FCF utilizes unirradiated values of yield and ultimate tensile stress to determine the stress limits based on Reference 15. The use of unirradiated values is conservative because irradiation has been shown to increase the yield and ultimate tensile stresses for Zircaloy. These criteria are consistent with the acceptance criteria established in Section 4.2 of the SRP and are acceptable up to the burnup limits established in Reference 4.

Evaluation - The stress analyses for FCF fuel assembly components and fuel rod cladding are based on standard stress analysis methods including finite-element analysis. Pressure and temperature inputs to the stress analyses are chosen so that the operating conditions for all normal operation and AOOs are enveloped. The cladding wall thicknesses are reduced to those minimum values allowed by fabrication specifications and further reduced to allow for corrosion on the inside and outside diameter. FCF uses the cladding corrosion from COROSO2 to determine corrosion on the outside diameter. PNNL concludes that the FCF design analysis methods for stress analyses are consistent with the guidelines in Section 4.2 of the SRP and are acceptable up to the burnup limits established in Reference 4.

(B) STRAIN

Bases/Criteria - The FCF design criteria for fuel rod cladding strain is that maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%. This criteria is intended to preclude excessive cladding deformation from normal operation and AOOs. This is the same criterion for cladding strain that is used in Section 4.2 of the SRP and, therefore, is acceptable.

The material property that could have a significant impact on the cladding strain limit at extended burnup levels is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operation, to levels that would allow cladding failure without the 1% cladding strain criteria being exceeded under normal operation and AOOs.

Recent out-of-reactor measured elastic and plastic cladding strain values from high burnup cladding from two PWR fuel vendors (References 16, 17 and 18) have shown a decrease in cladding ductilities when local burnups exceed 52 MWd/kgM. The cladding plastic strain values have a large scatter when local burnups were between 55 and 63 MWd/kgM with cladding ductility varying between 0.3% to 2% depending on testing methods (burst, tensile or ring tests), hydrogen levels in the cladding and fuel vendor. A quantitative separation of test methods and fuel vendor differences among the data is not possible at this time because of the large amount of scatter in the data and the relatively small amount of data at both high burnups and high corrosion levels. However, qualitatively the burst test data generally has the lowest cladding strains indicating that the stress state in the cladding appears to have some influence on measured uniform strain. Another complicating factor is that none of these testing methods, including the burst tests, simulate the stress state of pellet-cladding interaction (PCI) that contributes to cladding strain in operating fuel rods. However, all of these data do show that cladding ductility is decreasing with increasing burnup and hydrogen (corrosion) levels. In addition, the majority of the high burnup data (tensile or burst) shows that when hydrogen levels start to exceed 700 ppm the uniform strains begin to fall below 1%.

FCF has responded (Reference 8) with actual in-reactor strain data due to PCI above 1% strain without failure from segmented fuel rods ramped to peak powers of 12 to 13.4 kW/ft with peak burnups of 62 GWd/MTU and cladding hydrogen levels between 225 to 320 ppm (corrosion thickness between 39 to 55 microns). This demonstrates that the FCF cladding up to peak fuel burnups of 62 GWd/MTU can achieve elastic plus plastic strains of 1% or greater without failure but does not address FCF cladding ductility when hydrogen levels exceed 700 ppm. FCF's limit on maximum cladding corrosion (Reference 4) is consistent with maintaining cladding hydrogen levels below 700 ppm. PNNL concludes that the 1.0% uniform strain limit on FCF Zircaloy-4 cladding strain is acceptable up to the burnup and corrosion limits established in Reference 4.

Evaluation - The subject topical report has stated that the TACO-3 fuel performance code (Reference 19) is used for cladding strain analyses. This fuel performance code has been previously reviewed and approved by NRC up to the burnup levels established in Reference 4. FCF uses conservative bounding values for input to TACO-3 for this calculation including worst case fabrication tolerances, pressure differentials and power histories (including AOOs). PNNL concludes that this analysis methodology is acceptable.

(C) STRAIN FATIGUE

Bases/Criteria - The FCF design criterion for cladding strain fatigue is that the cumulative fatigue usage factor be less than 0.9 when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, which ever is the most conservative, is imposed as per the O'Donnell and Langer design curve (Reference 20) for fatigue usage.

The material property that could have a significant effect on the strain fatigue criterion is cladding ductility. As discussed in the above Section 3.0(B) for design strain, extended burnup operations above local burnups of 52 MWd/kgM have recently demonstrated a significant reduction in cladding ductilities. This could also reduce the cladding strain fatigue capability. However, as discussed in Section 3.0(B), Zircaloy-4 cladding ductility will not fall below the acceptable limit for total uniform strain if cladding corrosion and hydrogen levels are within the limits established by FCF in Reference 4. In addition, there is a considerable amount of conservatism in the FCF strain fatigue calculation and considerable lifetime margin in FCF strain fatigue results up to the burnup limits established in Reference 4. Also, the rod power for a FCF lead fuel rod at the extended burnup levels requested is relatively low so that cladding stress and strains will be relatively low at this burnup level. Therefore, PNNL concludes that the FCF strain fatigue criterion proposed in Reference 1 is acceptable for licensing applications to FCF fuel designs up to the burnup limits established in Reference 4.

Evaluation - The analysis methodology for evaluating strain fatigue for the FCF fuel designs uses the O'Donnell and Langer curve for irradiated Zircaloy (Reference 20). The use of O'Donnell and Langer's curve and analysis methods for determining strain fatigue life is consistent with SRP Section 4.2 and have been previously approved by the NRC. The analysis methodology also uses conservative inputs of minimum as-fabricated cladding thickness and oxide layer thickness. PNNL concludes that the strain fatigue analysis methods are acceptable for evaluating the above design criteria up to the burnup limits established in Reference 4.

(D) FRETTING WEAR

Bases/Criteria - Fretting wear is a concern for fuel, burnable poison rods, and guide tubes. Fretting, or wear, may occur on the fuel and/or burnable rod cladding surfaces in contact with the spacer grids if there is a reduction in grid spacing loads in combination with small amplitude, flow induced, vibratory forces. Guide tube wear may result when there is flow induced motion between the control rod ends and the inner wall of the guide tube.

While Section 4.2 of the SRP does not provide numerical bounding value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in the safety analysis report and that the stress/strain and fatigue limits should presume the existence of this wear.

The FCF design criterion against fretting wear is that the fuel design shall provide sufficient support to limit fuel rod vibration and cladding fretting wear. This design criterion can also be applied to other fuel assembly components that are susceptible to fretting wear, such as the fuel assembly guide tubes. This criterion is consistent with Section 4.2 of the SRP and is found to be acceptable for the FCF fuel designs up to the burnup levels established in Reference 4.

Evaluation - FCF has stated that fretting wear is based on external life and wear testing performed in a flow loop and postirradiation examination (PIE) results. The life and wear tests are conducted at maximum reactor flow conditions for more than 1000 hours to evaluate the fretting characteristics of the fuel rods and spacer grids.

FCF was questioned on the recent fretting failures in a FCF designed plant and whether this was due to irradiation induced relaxation of the spacer grid springs. FCF responded that the failures were from a non-FCF fuel design from another vendor and that some FCF spacer grid fretting problems had been observed in an old discontinued fuel design with Inconel intermediate spacer grids. They further indicated that two fretting failures have been found with their newer Zircaloy spacer grids but these were thought to be due to fabrication problems with the spacer grid springs or fuel handling had damaged the spacer grid springs in these two failure incidents. PNNL agrees that these are likely reasons for these fretting failures. FCF stated that they have examined Mark B designs with Zircaloy spacer springs up to very near the burnup limit in Reference 4 without any unusual observed fretting wear. Therefore, PNNL concludes that the evaluation of fretting wear has been adequately addressed up to the burnup limits established in Reference 4.

It should be noted that, recently, there have been more cladding fretting failures due to fabrication problems or flow anomalies from different vendors. These fretting failures have resulted in high plant coolant activities. In the future further NRC inspections may be required to examine this problem.

(E) OXIDATION AND CRUD BUILDUP

Bases/Criteria - Section 4.2 of the SRP identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. The SRP does not establish specific limits on cladding oxidation and crud but does specify that their effects be accounted for in the thermal and mechanical analyses performed for the fuel. As noted in Sections 3.0(B) and 3.0(C), the cladding ductility can be significantly decreased at higher burnup levels where oxide thickness and hydrogen levels can become relatively large because of accelerated corrosion at rod-average burnups above 50 to 55 GWd/MTU. FCF originally proposed a maximum corrosion limit that could achieve cladding hydrogen levels of 700 ppm and greater using the new FRAPCON-3 hydrogen pickup fraction due to corrosion (Reference 21). Due to the lack of strain data from FCF cladding with 700 ppm of hydrogen and above, FCF has revised their maximum corrosion limit (Reference 4) to be more consistent with existing hydrogen and strain data to date that demonstrates adequate cladding ductility. This maximum corrosion limit is based on a localized axial position on a fuel rod. PNNL concludes that this revised maximum corrosion limit (Reference 4) is acceptable up to the burnup limits established in Reference 4.

Evaluation - Section 4.2 of the SRP states that the effects of cladding crud and oxidation needs to be addressed in safety and design analyses, such as in the thermal and mechanical analysis. The amount of cladding oxidation is dependent on fuel rod powers, water chemistry control and primary inlet coolant temperatures, but the amount of oxidation and crud buildup increases with burnup and cannot be eliminated. Therefore, extended burnups result in a thicker oxide layer that provides an extra thermal barrier, cladding thinning and ductility decrease that can affect the mechanical analysis. The degree of this effect is dependent on reactor coolant temperatures and the level of success of a reactors' water chemistry program. The following is an evaluation of the FCF corrosion model.

FCF has proposed a new cladding corrosion model, COROSO2 (Reference 9), that is more conservative, i.e., predicts more corrosion, than the original OXIDEPC model in TACO3 and predicts the accelerated corrosion observed in high burnup rods much better than the OXIDEPC model. The relatively small amount of maximum corrosion thickness data from FCFs low tin cladding (currently the cladding used by FCF for high burnup applications) indicates that the COROSO2 model predicts maximum corrosion thickness in a best estimate or slightly conservative manner but significantly overpredicts span average corrosion (span average thickness is the type of data most often collected by FCF). It is the maximum corrosion thickness within an assembly or on a fuel rod that is of greatest interest for licensing analyses because this is the most likely point of failure due to corrosion and is the basis for the FCF corrosion limit discussed above. For this reason FCF plans to collect more data based on maximum corrosion thickness in the future. The maximum corrosion thickness measured by FCF is a moving average of the eddy current data over the rod length. The average is based on less than a half inch length of the fuel rod. The best estimate or slightly conservative prediction of the COROSO2 model is considered to be acceptable because of the conservatism in the FCF maximum corrosion limit. PNNL concludes that the COROSO2 model is acceptable for use in predicting maximum corrosion levels for verifying that they are within their proprietary maximum corrosion limit (Reference 4).

(F) ROD BOWING

Bases/Criteria - Fuel and burnable poison rod bowing are phenomena that alter the design-pitch dimensions between adjacent rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than place design limits on the amount of bowing that is permitted, the effects of bowing are included in the departure from nucleate boiling (DNB) analysis by a DNB ratio (DNBR) penalty when rod bow is greater than a predetermined amount. This FCF approach is consistent with the Section 4.2 of the SRP and is acceptable up to the burnup limits established in Reference 4.

Evaluation - Rod bowing has been found to be dependent on the distance between grid spacers, the rod moment of inertia flux distribution and material characteristics of the cladding. FCF has presented rod bowing data up to assembly average burnups of 58.3 GWd/MTU that shows that rod bowing saturates above 30 GWd/MTU and does not increase between 30 to 58.3 GWd/MTU. FCF has proposed an "observed limit" on rod bowing that bounds all of their data for use in their DNB analyses at rod-average burnups above 29 GWd/MTU. This "observed limit" is much greater than the 95% tolerance limit curve for their current data and, therefore, is conservative. FCF has further stated that the local power peaking uncertainties, used to accommodate rod bow effects, equal or bound the "observed limit" for assembly-average burnups greater than 29 GWd/MTU. PNNL concludes that this approach is conservative and, therefore, acceptable up to the burnup limits established in Reference 4.

(G) AXIAL GROWTH

Bases/Criteria - The FCF design basis for axial growth is that adequate clearance be maintained between the rod ends and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly. Similarly, for assembly growth, FCF has a design basis that axial clearance between core plates and the bottom and top assembly nozzles should allow sufficient margin for fuel assembly irradiation growth during the assembly lifetime to prevent the holddown spring in the assembly upper end fitting from going solid at cold shutdown. These criteria are consistent with Section 4.2 of the SRP and are acceptable up to rod-average burnup limits identified in Reference 4.

Evaluation - FCF provides an initial fuel rod-to-nozzle growth gap in their fuel assembly designs to allow for differential irradiation growth and thermal expansion between the fuel rod cladding and the fuel assembly guide thimble tubes. The minimum gap required to allow for the irradiation growth and thermal expansion to preclude interference during operation is based on the assumption of worst case fuel rod (maximum) and fuel assembly growth (minimum) combined with worst case fabrication tolerances. In like manner FCF designs holddown springs for the assembly to have enough travel to prevent the holddown spring from bottoming out on reactor-internals assuming maximum assembly growth and worst case tolerances.

The FCF models used to predict fuel rod and assembly growth are based on axial growth data for the Mark-B fuel design up to near the extended burnup limit requested for this design. FCF utilizes lower and upper bound 95/95 tolerance lines of their axial assembly growth data to predict the rod-to-nozzle gap and the assembly-to-reactor-internals gap to prevent the holddown spring from going solid, respectively. The upper bound 95/95 tolerance line for rod growth is used in the rod-to-nozzle gap analysis. Worst case fabrication dimensions or 95/95 dimensional tolerances (when available) are used in determining minimum gap spacings. PNNL concludes that these analysis methods are conservative and, therefore, are acceptable up to the burnup limits established in Reference 4.

(H) ROD INTERNAL PRESSURE

Bases/Criteria - Rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. Section 4.2 of the SRP presents a rod pressure limit of maintaining rod pressures below system pressure that is sufficient to preclude fuel damage. The FCF design basis for the fuel rod internal pressure is that the fuel system will not be damaged due to excessive fuel rod internal pressure and FCF has established the "Fuel Rod Pressure Criterion" (Reference 22) to provide assurance that this design basis is met. The internal pressure of the FCF lead fuel rod in the reactor is limited to a value below that which could cause 1) the diametral gap to increase due to outward cladding creep during steady-state operation, and 2) extensive DNB propagation to occur. This FCF design basis and the associated limits have been found acceptable by the NRC (Reference 22) up to the burnup limits established in Reference 4.

Evaluation - FCF utilizes the TACO3 fuel performance code (Reference 19) for predicting end-of-life (EOL) fuel rod pressures to verify that they do not exceed the FCF "Fuel Rod Pressure Criterion" during normal operation and AOOs. FCF was questioned (Reference 7) on the conservatism in the TACO3 code for predicting fission gas release (FGR) and rod pressures for steady-state and Condition 1 transients at extended burnups. FCF responded (Reference 8) by providing TACO3 predictions of two high burnup (62 GWd/MTU rod-average) fuel rods subjected to power ramps of 12 and 13.4 kW/ft and three high burnup (46 to 69 GWd/MTU) fuel rods operating at low steady state powers. The TACO3 code overpredicted the FGR of all five of these rods. A conservative power history is used by FCF in the EOL rod pressure analysis that includes several Condition 1 transients that bound any normal operation and AOOs. This application of power histories for the EOL rod pressure analysis were previously reviewed and approved (Reference 5) and are considered to be applicable to the burnups established in Reference 4. However, FCF has requested that a new power uncertainty factor derived from their newly approved neutronics methods based on the NEMO neutronics code (Reference 3) also be applied to their thermal-mechanical analyses including the EOL rod pressure analysis. PNNL concludes that the TACO3 analysis methods including the new power uncertainty factor from Reference 3 are applicable up to the burnup limits established in Reference 4.

(I) ASSEMBLY LIFTOFF

Bases/Criteria - The SRP calls for the fuel assembly hold-down capability (wet weight and spring forces) to exceed worst-case hydraulic loads for normal operation, which includes AOOs. The FCF assembly holddown criteria is "the holddown spring shall be capable of maintaining fuel assembly contact with the lower support plate during Condition I and II events." PNNL concludes that this is consistent with the SRP guidelines and, therefore, is acceptable up to the burnups established in Reference 4.

Evaluation - The fuel assembly liftoff forces are a function of primary coolant flow, holddown spring forces, and assembly dimensional changes. Extended burnup operation will result in additional irradiation relaxation of holddown springs and increase the fuel assembly length [assembly length changes are discussed in Section 3.0(G)]. These two phenomena have opposing effects on assembly holddown forces. For extended burnup operation the primary concern is that the holddown spring will go solid or increase spring forces to the point that fuel assembly bowing will occur and limit control rod insertion. Therefore, PNNL concludes that assembly liftoff is not a problem for FCF designs up to the burnup limits established in Reference 4.

4.0 FUEL ROD FAILURE

In the following paragraphs, fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the SRP will be reviewed. When the failure thresholds are applied for normal operation including AOOs, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of the GDC 10. When these thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose assessments required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus established by GDC 10 and Part 100 and only the threshold values and the analysis methods used to assure that they are met are reviewed below.

(A) HYDRIDING

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities in the fuel during fabrication; this is generally an early-in-life failure mechanism. FCF has not discussed their criteria for internal hydriding in the subject topical report; however, a limit on hydrogen level for FCF pellets is discussed in Reference 5. The hydrogen level of FCF fuel pellets is controlled by drying the pellets in the cladding and taking a statistical sample to ensure that the hydrogen level is below a specified level. Previous FCF design reviews, e.g., Reference 5, have shown that this level is below the value recommended in the SRP. Consequently, PNNL concludes that the FCF limit on hydrogen in their fuel pellets is acceptable.

External hydriding of the cladding due to waterside corrosion is the other source and is discussed in Section 3.0(E) of this TER. As noted in this section the level of external hydriding is controlled by FCF by a proprietary limit on corrosion thickness. PNNL concludes that this corrosion limit is acceptable for limiting the level of external hydriding in the cladding up to the burnup limits established in Reference 4.

Evaluation - Internal hydriding is controlled by FCF by taking statistical samples following pellet fabrication prior to loading the pellets in the fuel rods and confirming that hydrogen is below a specified level. Therefore, no analyses are necessary other than to confirm that the statistical pellet sampling is below the specified level.

External hydriding is controlled by the FCF limit on corrosion thickness discussed in Section 3.0 (E) of this TER.

PNNL concludes that FCF has addressed the issue of hydriding up to the burnup limits established.

(B) CLADDING COLLAPSE

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to fuel densification, the potential would exist for the cladding to collapse into a gap (i.e., flattening). Because of the large local strains that would result from collapse, the cladding is then assumed to fail. It is a FCF design criteria that cladding collapse is precluded during the fuel rod design lifetime. This design basis is the same as that in the SRP and. thus, is acceptable up to the burnup limits established in Reference 4.

Evaluation - The FCF analytical models for evaluating cladding creep collapse are the CROV and TACO3 computer codes that have been reviewed and approved by NRC (References 23 and 19). The application of these codes to calculating creep collapse are discussed in Reference 23. PNNL concludes that the application of these codes and methods are conservative for evaluating cladding creep collapse and, therefore, are acceptable up to the burnup limits established in Reference 4.

(C) OVERHEATING OF CLADDING

Bases/Criteria - The FCF design criteria for the prevention of fuel failures due to overheating is that there will be at least 95% probability, at a 95% confidence level, that DNB will not occur on a fuel rod during normal operation and AOOs. This design limit is consistent with the thermal margin criterion of the SRP guidelines and, therefore, is acceptable.

Evaluation - As stated in the SRP, Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit DNB or boiling transition in the core is satisfied. The principle physical phenomenon that is both burnup dependent and impacts DNB is fuel rod bowing and this is addressed in Section 3.0(F) of this report. This section demonstrates that rod bowing saturates at a burnup of 30 GWd/MTU and, therefore, DNB is not impacted up to the burnup levels established in Reference 4. PNNL concludes that FCF has addressed the issue of DNB.

(D) OVERHEATING OF FUEL PELLETS

Bases/Criteria - To preclude overheating of fuel pellets, FCF has indicated that no fuel centerline melting is allowed for normal operation and AOOs. This design limit is the same as given in Section 4.2 of the SRP and, therefore, is acceptable.

Evaluation - FCF was questioned about the recently observed reduction in fuel thermal conductivity reduction at extended burnups and its impact on TACO3 calculated fuel temperatures in relation to their fuel melt temperature analyses (Reference 7). FCF responded (Reference 8) that they evaluated the impact of the decrease in fuel thermal conductivity in TACO3 calculations based on both currently published information on the thermal conductivity decrease and previous TACO3 comparisons to fuel centerline temperature data up to a rod-average burnup of 40 GWd/MTU (Reference 19). FCF concluded that the TACO3 code provided a satisfactory prediction of fuel centerline temperature up to the burnup level that they had data (40 GWd/MTU rod-average), but because they had no data above this burnup level they would apply a penalty factor as a function of burnup above 40 GWd/MTU on TACO3 calculated fuel centerline temperatures for their fuel melting analyses. PNNL has evaluated FCF's methodology for developing and applying their penalty factor to TACO3 calculated fuel centerline temperatures for their fuel melting analyses. PNNL agrees that TACO3 provides an adequate prediction of fuel centerline temperature up to a rod-average burnup of 40 GWd/MTU and also finds that the penalty factor is satisfactory based on fuel thermal conductivity data available at this time.

Therefore, PNNL concludes that the new FCF penalty factor for fuel melting analyses, that accounts for the reduction in fuel thermal conductivity with burnup, is acceptable up to the burnup limits established in Reference 4.

(E) PELLET-CLADDING INTERACTION

Bases/Criteria - As indicated in Section 4.2 of the SRP, there are no generally applicable criteria for PCI failure. However, two acceptable criteria of limited application are presented in the SRP for PCI: 1) less than 1% transient-induced cladding strain, and 2) no centerline fuel melting. Both of these limits are used by FCF as discussed in Sections 3.0(B) and 4.0(D) of this report and, therefore, have been addressed by FCF.

Evaluation - As noted earlier, FCF utilizes the TACO-3 (Reference 19) code to show that their fuel meets both the cladding strain and fuel melting criteria. This code is acceptable per the recommendations in Sections 3.0(B) and 4.0(D).

(F) CLADDING RUPTURE

Bases/Criteria - There are no specific design limits associated with cladding rupture other than the 10 CFR 50 Appendix K (Reference 24) requirement that the incidence of rupture not be underestimated. A cladding rupture temperature correlation must be used in the loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) analysis. FCF uses a rupture temperature correlation consistent with NUREG-0630 guidance (Reference 25). PNNL therefore concludes that FCF has adequately addressed the criteria for cladding rupture.

Evaluation - FCF has adopted the cladding deformation and rupture models from NUREG-0630 guidance (Reference 25) which has been approved by the NRC for ECCS evaluation. The increase in fuel rod pressures with increasing burnup can impact cladding deformation and rupture. As noted in Sections 3.0(H) and 5.0(A) of this report, FCF uses the TACO3 fuel performance code to provide initial rod pressures and stored energy for the LOCA analysis and the code application of this code is found to be satisfactory for these applications up to the burnup levels established in Reference 4. PNNL concludes that FCF has adequately addressed the issue of cladding rupture.

(G) FUEL ROD MECHANICAL FRACTURING

Bases/Criteria - The term "mechanical fracture" refers to a fuel rod defect that is caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. The design limit proposed by FCF to prevent fracturing is that the stresses due to postulated accidents in combination with the normal steady-state fuel rod stresses should not exceed the yield strength of the components in their fuel assemblies. This design limit for fuel rod mechanical fracturing is consistent with the SRP guidelines, and, therefore, is acceptable.

Evaluation - The mechanical fracturing analysis is done as a part of the seismic-and-LOCA loading analysis. A discussion of the seismic-and-LOCA loading analysis is given in Section 5.0(D) of this TER.

5.0 FUEL COOLABILITY

For postulated accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). In the following paragraphs, limits and methods to assure that coolability is maintained are discussed for the severe damage mechanisms listed in the SRP.

(A) FRAGMENTATION OF EMBRITTLED CLADDING

Bases/Criteria - The most severe occurrence of cladding oxidation and possible fragmentation during a postulated accident is the result of a LOCA. In order to reduce the effects of cladding oxidation during a LOCA, FCF uses a limiting criterion of 2200°F on peak cladding temperature

(PCT) and a limit of 17% on maximum cladding oxidation as prescribed by 10 CFR 50.46. These criteria are consistent with SRP criteria and, thus, are acceptable.

Evaluation - FCF has stated that they will only use NRC reviewed and approved LOCA models for evaluating the above criteria. However, the initial fuel stored energy can impact the cladding embrittlement. FCF uses the TACO3 code to calculate initial stored energy for input to the LOCA analyses. FCF was questioned (Reference 7) about the impact on the calculated stored energy for LOCA due to the observed decrease in fuel thermal conductivity and the shift in radial power distributions at extended burnups because the TACO3 code does not accurately model these effects. FCF responded (Reference 8) that they have evaluated the impact of these effects on stored energy at extended burnup and propose to apply a penalty faster that increases their multiplicative uncertainty factors for TACO3 calculated stored energy to account for these effects. FCF further responded that the additional uncertainty factors would only be applied at burnups greater than 40 GWd/MTU (rod-average) because the TACO3 code has conservatisms built in that compensate for these effects below 40 GWd/MTU. This is demonstrated by the fact that the TACO3 code predictions and uncertainties have been shown to be satisfactory by comparison to fuel temperature data up to 40 GWd/MTU (rod-average). PNNL concurs with FCF that the conservatisms in TACO3 account for the effects of the thermal conductivity degradation and change in radial power distribution because the effects are small below 40 GWd/MTU. PNNL also concurs that the FCF proposed additional uncertainty factors on stored energy above 40 GWd/MTU (rod-average) are satisfactory based on the thermal conductivity data available at this time.

FCF has indicated that the LOCA analyses will continue to be limiting at beginning-of-life even with the use of these penalty factors above burnups of 40 GWd/MTU up to currently approved burnup levels. PNNL concludes that FCF has adequately addressed the impact of extended burnup on stored energy and LOCA up to the burnup levels established in Reference 4.

(B) VIOLENT EXPULSION OF FUEL

Bases/Criteria - In a severe reactivity insertion accident (RIA), such as a control rod ejection accident, large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal might be sufficient to destroy the fuel cladding and rod bundle geometry and to provide significant pressure pulses in the primary system. To limit the effects of an RIA event, Regulatory Guide 1.77 (Reference 26) recommends that the radially-averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g. In addition, the fuel failure limit is the onset of DNB for the close consequences of an RIA. The limiting RIA event for FCF fuel designs is a control rod ejection accident.

The FCF safety criteria for the control rod ejection accident is: the radial average peak fuel enthalpy for the hottest fuel rod shall not exceed

280 cal/g. This is identical to the guidance in Section 4.2 of the SRP and Regulatory Guide 1.77 (References 11 and 26). It is noted that the NRC staff are currently reviewing the 280 cal/gm limit and the limit for fuel failure may be decreased to a lower limit at high burnup levels. Recent RIA testing has indicated the fuel expulsion and fuel failure may occur before the 280 cal/gm limit and the onset of DNB, respectively (References 27 and 28). However, further testing and evaluation is needed to establish limits. The fuel expulsion and failure limits for an RIA may decrease in the future but the current limits remain valid at this time.

Evaluation - FCF verifies that this acceptance criterion is met for each fuel cycle through design and cycle specific analyses and by limiting the ejected rod worth. The industry and NRC have both done preliminary evaluatiooons of the worst impact of both a lower enthalpy limit for fuel expulsion and lower failure limit at current burnup limits are acceptable. The very conserfative analyses indicate that maximum enthalpies for high burnup rods are at least a factor of three lower than the current limit and violent expulsion is unlikely. The dose consequences are within those specified in 10 CFR 100. FCF uses NRC-approved methods to perform these analyses and the methods remain valid at this time up to the burnups established in Reference 4. PNNL concludes that FCF has adequately addressed this issue.

(C) CLADDING BALLOONING

Bases/Criteria - Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. There are no specific design limits associated with cladding ballooning other than the 10 CFR 50 Appendix K requirement that the degree of swelling not be underestimated. To meet the requirement of 10 CFR 50 Appendix K, the burst strain and the flow blockage resulting from cladding ballooning must be taken into account in the overall LOCA analysis. FCF has stated that they utilize the approved burst strain and flow blockage models developed from NUREG-0630 (Reference 25). It is noted that NRC is currently looking at the impact of the reduction in cladding ductility at extended burnups on cladding ballooning and rupture during LOCAs. However, the NUREG-0630 models remain applicable and valid at this time up to the burnup limits established in Reference 4.

Evaluation - FCF has adopted the cladding rupture and ballooning models from NUREG-0630 (Reference 25) as recommended by Section 4.2 of the SRP and these models have been previously approved by the NRC. Therefore, PNNL concludes that FCF has addressed the issue of cladding ballooning.

(D) FUEL ASSEMBLY STRUCTURAL DAMAGE FROM EXTERNAL FORCES

Bases/Criteria - Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. Appendix A to SRP Section 4.2 states that the fuel system coolable geometry shall be maintained and damage should not be so severe as to prevent control rod insertion during seismic and LOCA events. FCF has adopted the SRP guidelines as their design criteria. PNNL concludes that the FCF design criteria for seismic and LOCA loads are acceptable.

Evaluation - FCF stated that they have used NRC-approved methodologies provided in Reference 27 for evaluating seismic and LOCA loads. Extending fuel rod burnup levels could result in adverse effect on fuel assemblies due to seismic and LOCA events. FCF responded that the following parameters could impact seismic/LOCA events: spacer grid spring relaxation [see Section 3.0(D)], holddown spring relaxation, and reduction in the rod-to-nozzle and assembly-to-reactor-internals gaps [see Sections 3.0(G) and 3.0(I)], and changes in Zircaloy material properties [see Sections 3.0(A) and 3.0(B)]. FCF claims (Reference 1) that the relaxation of the spacer grid springs only decrease the natural frequencies of the assembly slightly based on post-irradiation-examination (PIE) data, and this small decrease has an insignificant effect on the spacer grid impact loads based on analysis studies. The reduction in the rod-to-nozzle and assembly-to-reactor-internals gaps are incorporated into the FCF dynamic response analysis for seismic/LOCA loads and the holddown spring relaxation has little effect because the spring rate is not affected (Reference 1). The change in material properties are primarily the increase in yield and ultimate tensile strength and the decrease in Zircaloy ductility. The increase in material strength results in greater assembly strength that is not accounted for by FCF for this analysis and, therefore, is conservative and acceptable. The reduction in Zircaloy ductility is controlled by the limit on corrosion discussed in Section 3.0(E) of this TER and, therefore, is acceptable. PNNL concurs that the extended burnup levels established in this TER will have an insignificant effect on seismic/LOCA loads. PNNL concludes that FCF has adequately addressed the issue of assembly loads due to seismic/LOCA.

6.0 FUEL SURVEILLANCE

FCF was questioned about what future fuel surveillance would be performed to justify operation for each of their fuel designs for future burnup extensions. FCF responded (Reference 4) that their lead assembly programs generally consist of four to eight fuel assemblies with varying levels of extended burnup operation. Each lead assembly will be subjected to PIE that varies depending on utility support but generally consists of fuel rod oxide and diameter, fuel ro and assembly bow, and assembly holddown spring height measurements. The guide tube and overall assembly condition are also visually examined. FCF further stated that the appropriate data would be submitted to NRC for review and approval prior to any extensions in burnup beyond the limits approved in this TER.

PNNL notes that the NRC may also request data on control rod drop times or drag tests for assembly burnups beyond current FCF burnup limits. In addition, the NRC may want to see rod drop test or drag test data for new fuel designs. This is because of the decrease in control rod drop times recently observed in some Westinghouse fuel designs/plants that have achieved high burnups. PNNL concludes that FCF has addressed the issue of fuel surveillance.

7.0 CONCLUSIONS

PNNL has reviewed the extended burnup request submitted in BAW-10186P and the responses to requests for additional information (RAIs) in accordance with the SRP, Section 4.2. PNNL concludes that topical report BAW-10186P is acceptable for licensing application for FCF Mark B, BW15 and BW17 designs up to the burnup levels of 62 GWd/MTU for the former and 60 GWd/MTU for the latter two desings. This approval does not include extended burnup operation of the FCF Mark-C fuel designs. In addition, FCFs request to apply the NEMO calculational uncertainty for use in TACO3 licensing analysis is also acceptable.

8.0 REFERENCES

1. Babcock and Wilcox Fuel Company. November 1992. Extended Burnup Evaluation. BAW-10186P, Babcock and Wilcox Fuel Company, Lynchburg, Virginia, transmitted by letter, J. H. Taylor (BWFC) to U.S. NRC Document Control Desk, "Submittal of Topical Report BAW-10186P," dated November 24, 1992.

2. Letter, J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC), dated July 19, 1995.

3. Babcock and Wilcox Fuel Company. March 1993. NEMO - Nodal Expansion Method Optimized. BAW-10180P-A. Rev. 1, Babcock and Wilcox Fuel Company, Lynchburg, Virginia.

4. Letter from J. H. Taylor (Framatome Cogema Fuels) to NRC Document Control Desk, "Application of BAW-10186P" JHT/97-7, dated January 23, 1997.

5. Babcock and Wilcox. April 1986. Extended Burnup Evaluation. BAW-10153P-A, Babcock and Wilcox, Lynchburg, Virginia.

6. Babcock and Wilcox Fuel Company. June 1988. Mark-BW Mechanical Design Report. BAW-10172P, Babcock and Wilcox Fuel Company, Lynchburg, Virginia.

7. Letter, Edward Kendrick (NRC) to J. H. Taylor (B&W Nuclear Technologies), "Request for Additional Information on Report BAW-10186-P, dated June 28, 1995.

8. Letter, J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC), JHT/95-88, dated August 22, 1995.

9. Letter, from J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (NRC), JHT/95-119, dated December 6, 1995.

10. Letter, from J. H. Taylor (Framatome Cogema Fuels) to R. C. Jones (NRC), JHT/96-042, dated June 26, 1996.

11. U.S. Nuclear Regulatory Commission. July 1981. "Section 4.2, Fuel System Design." In Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition. NUREG-0800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

12. United States Federal Register. "Appendix A, General Design Criteria for Nuclear Power Plants." 10 Code of Federal Regulations (CFR), Part 50. U.S. Printing Office, Washington, D.C.

13. United States Federal Register. "Reactor Site Criteria." 10 Code of Federal Regulations (CFR), Part 100. U.S. Printing Office, Washington, D.C.

14. United States Federal Register. 10 Code of Federal Regulations (CFR), Part 50.46. U.S. Printing Office, Washington, D.C.

15. American Society of Mechanical Engineers. 1983 Edition. "Section III, Nuclear Power Plant Components." ASME Code. American Society of Mechanical Engineers, New York.

16. Smith,Jr. G. P., R. C. Pirek and M. Griffiths. July 1994. *Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1*. EPRI TR-103302-V2, Final Report, Electric Power Research Institute, Palo Alto, California.

17. Newman, L. W. et al. 1986. *The Hot Cell Examination of Oconee Fuel Rods After Five Cycles of Irradiation*. DOE/ET/34212-50 (BAW-1874), Babcock & Wilcox, Lynchburg, Virginia.

18. Garde, A. M. 1989. "Effects of Irradiation and Hydriding on the Mechanical Properties of Zircaloy-4 at High Fluence." In *Zirconium in the Nuclear Industry: Eighth International Symposium*, ASTM STP 1023, pp. 548-569, Eds. L.F.P. VanSwam and C. M. Eucken. American Society for Testing and Materials, Philadelphia, Pennsylvania.

19. Wesley, D. A., and K. J. Firth. October 1989. TACO-3 Fuel Pin Thermal Analysis Code. BAW-10162P-A, Babcock & Wilcox, Lynchburg, Virginia.

20. O'Donnell, W. J., and B. F. Langer. 1964. "Fatigue Design Basis for Zircaloy Components." In Nuc. Sci. Eng. 20:1.

21. Lanning, D. D., C. E. Beyer, and C. L. Painter. March 1996. "New High Burnup Fuel Models for NRC's Licensing Code, FRAPCON." Proceedings of the U.S. Nuclear Regulatory Commission 23rd Water Reactor Safety Information Meeting, NUREG/CP-0249, Vol. 1, p. 141, Office of Nuclear Regulatory Research, U.S. NRC, Washington D.C.

22. D. A. Wesley, D. A. Farnsworth and G.A. Meyer, July 1995. Fuel Rod Gas Pressure Criterion (FRGPC), BAW-10183P-A.

23. Miles, T., et. al., August 1995. Program to Determine In-Reactor Performance of B&W Fuel Cladding Creep Collapse, BAW-10084P-A Rev.3, B&W Fuel Company, Lynchburg, Virginia.

24. United States Federal Register. "Appendix K, ECCS Evaluation Models." 10 Code of Federal Regulations (CFR), Part 50. U.S. Printing Office, Washington, D.C.

25. Powers, D. A., and R. O. Meyer. 1980. Cladding Swelling and Rupture Models for LOCA Analysis. NUREG-0630, U.S. Nuclear Regulatory Commission, Washington, D.C.

26. U.S. Nuclear Regulatory Commission. 1974. Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,
Regulatory Guide 1.77, U.S. Nuclear Regulatory Commission, Washington D.C.

27. Schmitz, F., et. al., March 1996. " New Results from Pulse Test in the CABRI Reactor," Proceedings of the 23rd Water Reactor Safety Information Meeting October 23-25,1995.

28. Fuketa, T., et. al., March 1996. "New Results from the NSRR Experiments with High Burnup Fuel," Proceedings of the 23rd Water Reactor Safety Meeting, October 23-25, 1995.

29. Babcock and Wilcox. May 1979. Analytical Methods Description - Reactor Coolant System Hydrodynamic Loadings During a Loss-of-Coolant-Accident - Supplement 1, BAW-10132P-A Supp.1, Babcock and Wilcox, Lynchburg, Virginia.